June 11, 1999

Mr. R. P. Powers
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: D. C. COOK INSPECTION REPORT 50-315/99010(DRP); 50-316/99010(DRP)

Dear Mr. Powers:

On May 27, 1999, the NRC completed an inspection at your D. C. Cook Units 1 and 2 reactor facilities. The inspection was an examination of activities conducted under your license as they relate to compliance with the Commission rules and regulations and with the conditions of your license. Areas reviewed included Operations, Maintenance, Engineering, and Plant Support. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations of activities in progress. The enclosed report presents the results of that inspection.

Your staff successfully restored the Unit 2 AB diesel generator to an operable condition along with the associated train of boration control resulting in an increased ability to address postulated reactor coolant dilution events. In addition, the continued effort to place both units on electrical back feed indicated an increased focus on reactor core safety issues. We also noted that during the conduct of the Expanded System Readiness Review process, your staff demonstrated a critical questioning attitude by effectively identifying technical issues related to the Containment Spray System.

While an increased focus on reactor core safety issues was evident, several examples were identified by the inspectors where your staff failed to consistently respond in a manner commensurate with the potential impact of the degraded condition on current plant configuration. The examples included your staffs' response to degraded and inoperable Essential Service Water pump discharge strainers, degraded 4 kV breakers, and source range instrument issues. In response to the inspectors' concerns, your staff took prompt and appropriate actions, including the initiation of compensatory measures when warranted. These actions ensured that equipment which could be required to support maintaining the reactor in a cold shutdown condition were available or returned to service in an expeditious manner.

During the inspectors review of previously identified regulatory issues, two violations of NRC requirements were identified. The first violation, identified by the inspectors in 1998, was due to an inappropriate surveillance procedure for the engineered safeguards ventilation system. The surveillance procedure did not direct the operators to restore the system to an operable configuration. The second violation, identified by your staff in 1995, involved the failure to restore several main steam safety valves to operable status prior to exceeding the Technical Specification Limiting Condition for Operation time limit.

These Severity Level IV violations are being treated as Non-Cited Violations (NCVs). Appendix C of the Enforcement Policy requires that for Severity Level IV violations to be dispositioned as NCVs, they be appropriately placed in the licensee’s corrective action program. Implicit in that requirement is that the corrective action program be fully acceptable. The plant corrective action program was not adequate and has been the focus of significant attention by your staff to improve the program. While your staff and the NRC have not yet concluded that the corrective action program is fully effective, the corrective action program improvement efforts are underway and captured in the Restart Plan which is under the formal oversight of the NRC through the NRC Manual Chapter 0350 Process, “Staff Guidelines for Restart Approval.” Consequently, these issues will be dispositioned as NCVs.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

Sincerely,

/s/ J. A. Grobe

John A. Grobe, Director
Division of Reactor Safety

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74

Enclosure: Inspection Report 50-315/99010(DRP);
50-316/99010(DRP)

cc w/encl: A. C. Bakken III, Site Vice President
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/99010(DRP); 50-316/99010(DRP)
Licensee:	Indiana and Michigan Power 500 Circle Drive Buchanan, MI 49107-1395
Facility:	Donald C. Cook Nuclear Generating Plant
Location:	1 Cook Place Bridgman, MI 49106
Dates:	April 17, through May 27, 1999
Inspectors:	B. L. Bartlett, Senior Resident Inspector B. J. Fuller, Resident Inspector J. D. Maynen, Resident Inspector K. A. Coyne, Project Engineer - Region II
Approved by:	A. Vegel, Chief Reactor Projects Branch 6 Division of Reactor Projects

EXECUTIVE SUMMARY

D. C. Cook Units 1 and 2
NRC Inspection Report 50-315/99010(DRP); 50-316/99010(DRP)

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 6-week period of resident inspection activities and includes follow-up to issues identified during previous inspection reports.

Operations

C Overall, plant operations were performed using approved operating procedures and reflected good operating practices. The inspectors observed an operator conduct in-plant rounds and determined that the operator was knowledgeable of system status and operation, observed proper radiological controls, and appropriately communicated with the control room. (Section O1.1)

C During a routine tour of the control rooms, the inspectors determined that the control room operators were not fully cognizant of the status of the configuration of the source range nuclear instruments. The source range instrument alarms were treated inconsistently between the units and the licensed operators were not knowledgeable of the differences or of the reasons for the differences. (Section O1.2)

C Several examples were identified by the inspectors where the licensee failed to consistently respond in a manner commensurate with the potential impact of the degraded condition on the current plant operational condition. The examples included degraded and inoperable Essential Service Water pump discharge strainers, and degraded 4 kV breakers and source range nuclear instruments. Licensee management personnel recognized the need to prioritize degraded material condition issues in order to focus on items important to reactor core safety in Mode 5, but did not always ensure the corrective actions were focused towards Mode 5 items first. (Section O1.2)

Maintenance

C The inspectors concluded that the maintenance and surveillance activities observed were performed in accordance with procedures. The current revision of the appropriate procedures were in use at the work sites, and appropriate radiological protection practices were noted. (Section M1.1)

• Plant material condition continued to be identified including problems with the ESW pump discharge strainers. The licensee's response to the degraded ESW strainers lacked a sense of urgency commensurate with their importance to maintaining reactor core safety. Following NRC inspector observations on the need to re-evaluate the corrective actions on the ESW strainers, the licensee's corrective actions were appropriate. (Section M2.1)

Engineering

C The inspectors observed that engineering personnel provided effective support to maintenance personnel in an effort to resolve the containment spray system problems in a timely manner. As appropriate, the problem scopes were expanded and/or re-assessed to ensure adequate technical resolutions were achieved. (Section E7.1)

- Significant progress had been made by the licensee in the identification of technical issues related to the Containment Spray System. At the end of the inspection report period, the system remained inoperable pending resolution of the identified issues. Case Specific Checklist item 10 remained open at the end of the inspection period. (Section E7.1)

Plant Support

- Delay in the placement of plant equipment into an appropriate lay-up condition contributed to the degradation of some plant safety and nonsafety-related equipment. The licensee had recognized the need for an effective lay-up program and had taken action to place equipment in lay-up and to develop planning for lay-up during future outages. (Section R2.1)

Report Details

Summary of Plant Status

The licensee maintained both Unit 1 and Unit 2 in Mode 5, Cold Shutdown, throughout the inspection period. During this inspection period, the licensee continued with the Expanded System Readiness Reviews (ESRR). The Unit 2 AB emergency diesel generator was returned to an operable status following HFA relay calibrations, which allowed the licensee to declare the Unit 2 west charging pump operable. In addition, work continued on restoring the other emergency diesel generators and the chemical and volume control system to an operable status. The licensee informed the NRC on April 30, 1999, that Unit 2 was the lead unit for restart work.

The licensee was pursuing placing both units on electrical back feed through the main transformers to improve connectivity to off-site power supplies. Various technical and material condition issues with the main transformers required resolution prior to placing the units on back feed.

I. Operations

O1 Conduct of Operations

O1.1 General Comments

The inspectors conducted frequent observations of control room and in-plant operation of equipment during the extended outage of both reactor units. Overall, plant operations were performed using approved operating procedures and reflected good operating practices. Specific events and noteworthy observations are detailed in the sections below.

On May 11, 1999, the inspectors questioned the licensee about which deficiencies would require correction to support Unit 2 entry into Mode 4 (reactor coolant system temperature greater than 200^{o}F but less than 350^{o}F). The licensee stated that a comprehensive list of Unit 2 Mode 4 constraints did not exist and that the current tracking processes did not capture all Mode 4 constraints. Condition Report 99-09835, written on April 29, 1999, had already documented this problem. The inspectors identified several Unit 1 Technical Specifications (TSs) required to be satisfied when Unit 2 was in Modes 1, 2, 3, or 4. These Unit 1 TS requirements were not tracked on the Unit 2 TS open items index. The licensee stated that the Unit 1 TS required equipment would also be added to the Unit 2 Mode 4 constraints in response to the inspectors comments.

On May 26, 1999, the inspectors accompanied a non-licensed operator on the Unit 2 Auxiliary Building Tour conducted in accordance with procedure 02-OHP [Operations Head Procedure] 5030.001.001, “Operations Plant Tours,” Revision 14. The operator was knowledgeable of system status and operation, observed proper radiological controls, appropriately communicated with the control room, and conducted a thorough tour. The inspectors observed minimal amounts of debris and transient combustibles in the auxiliary building. The inspectors noted that the reference information section of the operator rounds procedure incorrectly referenced the normal system condition of the 2-SYS3-NESW [Non-Essential Service Water] fire detection panel. However, the inspectors observed that the operator appropriately verified the condition of the 2-SYS3-NESW panel. The licensee initiated Condition Report (CR) 99-13653 to document this minor procedure deficiency.

O1.2 Licensee Focus on Reactor Core Safety

a. Inspection Scope (71707)

The inspectors identified several technical issues and noted that the licensee's initial response to these issues lacked a sense of urgency commensurate with managements expectations that corrective actions should be prioritized based on the potential impact on core safety. The inspectors reviewed the licensee's follow up to the recently identified technical issues and previously documented issues in order to perform an assessment of the licensee's focus on reactor core safety.

b. Observations and Findings

Since the licensee shutdown both units on September 8, 1997, various licensee efforts to enhance reactor core safety have been evident. Within the last 6 months, the licensee's improved focus on reactor core safety resulted in increased attention to emergent issues related to maintaining the core in a safe shutdown condition. Examples included the decision to perform the initial Expanded System Readiness Reviews (ESRR) on those systems necessary for reactor core safety in Mode 5, the efforts to increase electrical connections from the grid by going on main transformer back feed, the establishment of a core safety priority safety list, and the efforts to restore the emergency diesel generators and boron injection flow paths to an operable status.

Despite the improved performance, the NRC inspectors identified three examples where the licensee demonstrated an inadequate sense of urgency in prioritizing corrective actions to resolve issues involving degraded equipment. The first example involved the poor material condition of the Essential Service Water (ESW) pump discharge strainers, the second example involved the lack of a comprehensive preventive maintenance program on 4 kilovolt breakers, and the third example involved operator knowledge of the condition of the reactor vessel source range instruments. The licensee's initial response to correct these examples lacked a sense of urgency commensurate with the function of the affected systems in the current mode of operation.

b.1. ESW Pump Discharge Strainers

The licensee identified several operability and degraded material condition issues on all four (two per unit) ESW strainers. As issues were identified, the licensee performed operability evaluations as necessary for each individual issue and concluded that the ESW system remained operable. (The specific operability and material condition issues are discussed below in Section M2.1.) However, an aggregate operability evaluation of these issues was not performed until after questions had been raised by the inspectors. Although the Technical Specifications for ESW did not require the system to be operable in Mode 5, the ESW system was required as a support system for RHR, Component Cooling Water, and the emergency diesel generators which are systems required in Mode 5.

After the inspectors questioned the licensee's response to the questions of operability and material condition, the licensee:

C increased operator tours of the ESW system,

C tracked ESW strainer differential pressures,

C made contingency plans for the failure of an additional ESW strainer,

C reviewed loss of ESW abnormal operating procedures,

C reviewed the performance of strainer manual backwash procedure,

C requested vendor assistance, and

C performed an operability determination for the overall system degradation in order to ensure the ESW system could continue to support the removal of core decay heat.

Though the licensee was prompted by the inspectors, the licensee's subsequent response and corrective actions were adequate to compensate for the degraded material condition of the ESW system.

b.2. Safety-Related 4 Kilovolt (kV) Breakers

The licensee had identified that their 4 kV breakers were in a degraded material condition due to an inadequate preventive maintenance program. In response, an NRC inspection was performed and documented in Inspection Report 50-315/316-99011. As documented in that report, the licensee did not prioritize their refurbishment program to consider those breakers necessary to operate equipment needed for Mode 5, until after questions had been raised by the NRC inspectors.

b.3. Source Range Nuclear Instruments

During routine control room walkdowns, the inspectors observed that all four (two per unit) normal source range instrument channels were operable, and that all post accident source range detectors (referred to as the Gamma-Metrics detectors) had their high flux alarms disabled except for Unit 2 detector 2-NRI-21. Source range instruments were required for reactor core safety in order to monitor the shutdown status of the reactor core. The inspectors interviewed the unit supervisors and I&C personnel and determined that the normal licensee practice was to enable the Gamma-Metrics source range detector alarms only when the normal source range instruments were inoperable.

Unit 2 had experienced problems with their source range instruments in early February 1999, and the operators had enabled the alarm on Gamma-Metrics source range detector 2-NRI-21. Following the restoration of the normal source range instruments, the high flux alarm had not been disabled on 2-NRI-21.

The inspectors determined that the licensed operators were unaware of the status of the high flux alarms, the reasons for the current status of the alarms, or the desired alarm status. Even though the safety significance of the inconsistent enabling of the ex-core source range detectors high flux alarm was minimal, the occurrence of this issue indicated that the operators were not fully cognizant of the configuration of plant equipment, in this case the source range instruments.

Additional issues previously identified by the NRC inspectors included:

C The failure of licensed operators to understand which controlling procedures they were utilizing for maintaining the plant in a cold shutdown condition (Inspection Report 50-315/316-98008, Section O1.2).

C The failure of the licensed operators to know the time necessary for the core to reach boiling conditions in the event of a loss of all core cooling (Inspection Report 50-315/316-98016, Section O1.2).

C The failure of the licensed operators to identify erratic flow indications in the Unit 1 Residual Heat Removal (RHR) system. Following the inspectors' identification of the erratic flow, licensee personnel performed a detailed walk down of the RHR system and identified flow cavitation issues (Inspection Report 50-315/316-99001, Section O2.1).

C The failure of the licensed operators to recognize that reactor coolant system temperature changes while in cold shutdown also resulted in reactivity changes (Inspection Report 50-315/316-99004, Section O1.2).

C The failure of the licensee to restore fencing and other barriers around the Unit 1 condensate storage tank and the Unit 1 refueling water storage tank (RWST) following removal for the moving of the spare main transformer (Inspection Report 50-315/316-99004, Section O1.1).

In public Manual Chapter 0350 meetings, licensee and NRC personnel had discussed plant material condition. It had been recognized that efforts to identify the extent of the design and licensing issues would cause resources for corrective maintenance to be below normal levels and that this reduced resource availability would impact negatively upon plant material condition. The NRC had also discussed the importance of the licensee monitoring plant material condition and responding promptly to degraded and potentially non-conforming equipment issues that impacted equipment necessary for reactor core safety. The issues identified by the NRC in this inspection period showed that the licensee efforts in this area have been inconsistent.

b.4. Corrective Actions

The licensee initiated condition reports for the ESW strainer issues (CR 99-13076) and the control of the source range instrument high flux alarm set point (CR 99-14123). Resources were increased on the repair of the Unit 2 West strainer, and corrective actions were planned to address the specific material condition issues on all four ESW strainers. The 4 kV breakers had a root cause analysis performed and a vendor was brought on site to refurbish breakers. Additional investigations and corrective actions related to other licensee electrical component issues were continuing. The issues listed above that were discussed in other inspection reports had previously been addressed in CRs.

c. Conclusions

During a routine tour of the control rooms, the inspectors determined that the control room operators were not fully cognizant of the status of the configuration of the source range instruments. The source range instrument alarms were treated inconsistently between the units and the licensed operators were not knowledgeable of the differences or of the reasons for the differences.

Several examples were identified by the inspectors where the licensee failed to consistently respond in a manner commensurate with the potential impact of the degraded condition on the current plant operational condition. The examples included degraded and inoperable Essential Service Water pump discharge strainers, degraded 4 kV breakers, and source range nuclear instruments. Licensee management personnel recognized the need to prioritize degraded material condition issues in order to focus on items important to reactor core safety in Mode 5, but did not always ensure the corrective actions were focused towards Mode 5 items first.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726 and 62707)

The inspectors performed routine observations of maintenance and surveillance activities in progress. The inspectors observed procedure use and adherence, and radiological control practices.

b. Observations and Findings

The inspectors observed all or part of the following maintenance activities:

- 01-OHP [Operations Head Procedure] 4030.STP [Surveillance Test Procedure].027AB, AB Diesel Generator Operability Test (Train B), Revision 14

- 02-OHP 4030.STP.027AB, AB Diesel Generator (D/G) Operability Test (Train B), Revision 12

- Job Order (JO) C0048865, Unit 2 AB Emergency Diesel Generator, replace damaged oil line at 3R cylinder

- JO C0047947, Perform checks of HFA relays on Unit 2 AB D/G

- JO R0073261, Clean oil/air separator and crankcase blower

- JO R0084700, Sample and change the oil in the governor to Unit 2 AB D/G

- Action Request (AR) A179380, Unit 2 East ESW Supply fan number 1 has higher vibrations than the other fans.

- AR A183629, Unit 2 West ESW strainer lost power, perform troubleshooting activities.

- AR A183169, Unit 2 East ESW strainer basket backwash valve 2-WRV-773 operates in a "jerky" fashion.

The inspectors concluded that the observed work was performed in accordance with procedures. The current revision of the appropriate procedures were in use at the work sites, and proper work safety and radiological protection practices were noted. Inspector observations of work activities associated with the ESW pump discharge strainers are documented in Section M2.1.

M2 Material Condition of Facilities and Equipment

M2.1 Essential Service Water Pump Discharge Strainers Degraded Material Condition

a. Inspection Scope (62707 and 71707)

Plant operation and maintenance activities were impacted by the material condition of the plant. The inspectors followed up on several activities which were affected. The licensee's responses to material condition issues on various system components were discussed in Section O1.2, above. The inspectors assessed the material condition of the ESW pump discharge strainers as part of the continued evaluation plant material condition.

b. Observations and Findings

Each of the two Units has two ESW pumps. On the discharge of each pump is a duplex strainer. During normal operation, the strainer uses nonsafety-related equipment and based on a set differential pressure, automatically swaps sides and initiates a backwash of the side that had previously been in service. Previous NRC inspection reports identified that the licensee had considered the ESW strainers as a support system not required for operability of the ESW system (Inspection Report 50-315/316-96007) and a Notice of Violation had been issued for the licensee's failure to have procedures for the performance of manual back washing of the ESW system (Inspection Report 50-315/316-97024). In response to questions raised during the ongoing Expanded System Readiness Reviews (ESRRs), the licensee's position on the need for the strainers was being re-evaluated, but had not yet formally changed. As material condition issues on the ESW strainers were identified by the licensee, the issues were evaluated, and the importance of the strainers for support of the systems necessary for the safe shutdown operation of the units were recognized.

The licensee identified several operability and degraded material condition issues on the ESW strainers in addition to known, pre-existing issues already identified on the strainers. As the issues were identified, the licensee performed operability evaluations as necessary for each individual issue and maintenance personnel began corrective maintenance activities. However, an aggregate operability evaluation was not performed until after questions had been raised by the inspectors. Although the Technical Specifications for ESW did not require the system to be operable in Mode 5, the ESW system was required as a support system for RHR, component cooling water, and the emergency diesel generators.

On May 19, 1999, the Unit 2 West ("B" Train) strainer failed to complete a timed swap and backwash. At the time the Unit 1 East ("A" Train) strainer was out of service for troubleshooting and additional corrective maintenance for a previous failure of the inlet gate to operate. The Unit 1 East strainer was restored to an available status and troubleshooting begun on the Unit 2 West Strainer.

Other operability or material condition issues existing on the four ESW strainers at that time included:

- Failure of the Unit 1 West left strainer to backwash due to a failed backwash valve
- Degraded gate seal to the inlet gate of the Unit 1 East strainer
- Rounded key on the motor operator to the inlet gate of the Unit 1 East strainer
- Cracked support pads for both of the Unit 1 strainers
- Improperly supported air lines to the backwash valves of all four ESW strainers
- Improperly supported instrument lines to all four ESW strainers

- Jerky operation of Unit 2 East ESW strainer basket backwash valve 2-WRV-773

Following the inspectors' observations that there was a lack of a sense of urgency to initiate corrective actions and compensatory measures, commensurate with the importance to safety, the licensee significantly increased the focus and resources on the ESW strainers. There had been discussions by members of the licensee's management on the need to increase attention on the material condition and to initiate precautions such as discussed in the Section O1.2 above, but these discussion had not resulted in actions.

Condition Reports related to the ESW strainers were listed in Section O1.2. Corrective actions included:

- Regular focus meetings between maintenance supervisors, engineers, schedulers, operations department representatives, and the shift manager to track, trend, discuss, and resolve ESW strainer issues.

- Meetings with a representative of the ESW strainer vendor to resolve questions on the causes of the degraded material conditions and the necessary corrective actions. For example, the bolts holding the strainer slide gates to the operating shafts had backed out, loosened, or failed on the Unit 2 West strainer inlet and outlet gates. The licensee held discussions with vendor representatives to determine the need for longer bolts, the types of bolts, the types of nuts, torque values, and the advisability of using locking compound during re-installation.

- A schedule was drafted for the performance of corrective maintenance on all ESW strainers in order to return them to their original design conditions. At the end of the assessment period, the schedule was still in the planning stage.

- Maintenance procedures were being reviewed in order to determine necessary modifications, additions, and deletions.

c. <u>Conclusions</u>

Plant material condition continued to decline as evidenced by the licensee identified problems with the ESW pump discharge strainers. The licensee's response to the degraded ESW strainers lacked a sense of urgency commensurate with their importance to maintaining reactor core safety. Following NRC inspector observations on the need to re-evaluate the corrective actions on the ESW strainers, the licensee's corrective actions were appropriate.

M3 Maintenance Procedures and Documentation

M3.1 <u>(Closed) EEI 50-315/98021-03</u>: Surveillance procedure did not direct the operators to restore the AES system to a configuration included in the normal operating procedure or enter the appropriate TS limiting condition for operation action statement. On October 28, 1998, the inspectors identified that Surveillance Procedure **01-OHP 4030.STP.025A, "Engineered Safety Features Fan No. 1 (1-HV-AES-1) Ventilation

Exhaust Air Filter Train Test," Revision 3, was inappropriate to the circumstances in that it did not direct the operators to restore the AES system to a configuration included in the normal operating procedure or enter the appropriate TS limiting condition for operation action statement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings, required, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The inspectors considered the failure of the surveillance procedure to return the AES system to an operable lineup to constitute an apparent violation of 10 CFR Part 50 Appendix B, Criterion V. This apparent violation remained open to allow the licensee a reasonable time to develop corrective actions. On December 22, 1998, the licensee completed a review of the AES surveillance procedures. In addition to the deficiency described above, Condition Report 98-08328 documented other apparent deficiencies in the AES surveillance procedures. The licensee planned to revise the procedures to correct the identified problems.

This Severity Level IV violation is being treated as a Non-Cited Violation (NCV). Appendix C of the Enforcement Policy requires that for Severity Level IV violations to be dispositioned as NCVs, they be appropriately placed in the licensee's corrective action program. Implicit in that requirement is that the corrective action program be fully acceptable. The plant corrective action program was not adequate and has been the focus of significant attention by your staff to improve the program. While your staff and the NRC have not yet concluded that the corrective action program is fully effective, the corrective action program improvement efforts are underway and captured in the Restart Plan which is under the formal oversight of the NRC through the NRC Manual Chapter 0350 process, "Staff Guidelines for Restart Approval." Consequently, this issue will be dispositioned as an NCV (50-316/99010-01(DRP)). This EEI is closed.

M8 Miscellaneous Maintenance Issues

M8.1 <u>(Closed) Licensee Event Report 50-315/95002-00</u>: Nine Unit 1 main steam safety valves had lift setpoints above the Technical Specification allowed values. In June 1995, with the unit at approximately 55 percent power, the licensee exercised the Unit 1 main steam safety valves (MSSVs). Nine MSSVs were found to be above the 3 percent lift setpoint tolerance allowed by Technical Specification (TS) 3.7.1.1. These valves were not reset to within 1 percent of the nominal lift setpoint until after the setpoints were questioned by the inspectors. This event was discussed in Inspection Report 50-315/95009. Technical Specification 3.7.1.1 required, in part, that a safety valve shall be reset to the nominal value ± 1 percent whenever found outside the ± 1 percent tolerance. Contrary to the above, the licensee did not reset the MSSV setpoints to within 1 percent of the nominal lift setpoint until after questioned by the inspectors. The inspectors considered the failure to reset the MSSV lift setpoints to constitute a violation of TS 3.7.1.1. The licensee implemented more frequent MSSV exercising, and valve seat refurbishment on MSSVs found out of tolerance. Condition Report 98-07823 was initiated to address MSSV testing and refurbishment issues.

This Severity Level IV violation is being treated as an NCV. Appendix C of the Enforcement Policy requires that for Severity Level IV violations to be dispositioned as

NCVs, they be appropriately placed in the licensee's corrective action program. Implicit in that requirement is that the corrective action program be fully acceptable. The plant corrective action program was not adequate and has been the focus of significant attention by your staff to improve the program. While your staff and the NRC have not yet concluded that the corrective action program is fully effective, the corrective action program improvement efforts are underway and captured in the Restart Plan which is under the formal oversight of the NRC through the NRC Manual Chapter 0350 process, "Staff Guidelines for Restart Approval." Consequently, this issue will be dispositioned as an NCV (50-315/99010-02(DRP)). This LER is closed.

M8.2 (Closed) Licensee Event Report 50-315/97004-00: Main steam safety valve exceeds allowable lift setpoint due to setpoint drift. On February 28, 1997, during main steam safety valve (MSSV) testing, the licensee identified that MSSV 1-SV-2A-3 exceeded its allowable lift setpoint by greater than three percent. The licensee's investigation determined that the cause of the setpoint change was setpoint drift. The inspectors reviewed the licensee's logs and determined that the valve lift setpoint was reset and that the MSSV was restored to operable within 4 hours. The Limiting Condition for Operation for Technical Specification 3.7.1 was met; therefore, no violation of Technical Specifications occurred. Condition Report 98-07823 was initiated to address MSSV testing and refurbishment issues. This LER is closed.

III. Engineering

E1 Conduct of Engineering

E1.1 General Engineering Comments

The licensee's staff continued with the discovery phase of the ESRRs. The ESRRs comprised a significant portion of the licensee's restart effort. In accordance with the NRC Inspection Manual Chapter 0350 inspection plan, the NRC established an inspection team to provide oversight of the ESRR process. The NRC oversight team will document their findings in a separate inspection report.

E7 Quality Assurance in Engineering Activities

E7.1 Case Specific Checklist Item 10, "Resolution of Containment Spray System Operability Issues"

a. Inspection Scope (37551 and 71707)

During this inspection period the inspectors performed routine follow up on Manual Chapter 0350, Case Specific Checklist (CSC) Item 10, regarding resolution of the Containment Spray System (CTS) Operability Issues. These operability issues were identified by NRC, third party, and licensee personnel. The issues were documented in Licensee Event Report 50-315/980020, 022, 027, 030, and 034; Inspection Reports 50-315/98004, 005, and 009 and in numerous licensee Condition Reports (CRs). At the end of this assessment period, CSC item 10 remained open.

b. <u>Observations and Findings</u>

The licensee's efforts to identify and correct the CTS operability issues were focused primarily on three different areas:

- The rotation of the Unit 1, West CTS heat exchanger to its correct configuration and the problems identified during this evolution.
- The ESRR findings including the incorporation of NRC and third party assessments.
- The ESRR findings from other system assessments that impacted upon the CTS system.

The licensee's ESRR on the CTS system had recently been completed and the corrective actions were still being developed and prioritized at the end of this inspection report period. The findings by the ESRR on the CTS system included the following items:

- CR 99-5921: During a walk down of the CTS system, the ESRR team observed that the top 5 percent of the containment was not covered by the CTS sprays. This was contrary to an assumption used by a vendor to calculate the off-site and control room dose calculations.
- CR 99-5939: During a walk down of the CTS system, the ESRR team observed that several CTS upper containment nozzles had partially obstructed spray patterns from nearby lighting or lighting supports.
- CR 99-5934: During a walk down of the CTS system, the ESRR team observed that the fall height of the CTS spray in lower containment was significantly shorter than assumed by a vendor in dose calculations.
- CR 99-6013: During a walk down of the CTS system, the ESRR team observed that the spray pattern of the CTS was partially blocked by the licensee's practice of parking the polar crane above the air lock doors.

The inspectors have generally observed that the findings from the ESRR were indicative of a detailed assessment by personnel with good questioning attitudes. The inspectors will perform additional assessments of the problems identified on the CTS system during continued reviews of the licensees corrective actions in regard to the issues tracked by CSC item 10.

Findings by other ESRR teams that impacted upon the operability of the CTS included:

- CR 99-6389: The two CTS spray additive tank outlet valves on each unit were to have their actuator torque switches replaced in response to a 10 CFR Part 21 notification. The ESRR identified that the work had been deferred for five years, without adequate justification.

- CR 99-6012: An ESRR team identified that non-seismic class I lighting and supports were above the refueling canal drains that lead to the recirculation sump. During a postulated accident, the lights could fail and clog the drains potentially reducing the return of water to the recirculation sump.

- CR 99-7853: Two of the upper containment pressure transmitters and two of the lower containment pressure transmitters were identified as improperly mounted. The pressure transmitters were anchored to both the containment and the auxiliary buildings. Differential movement of the buildings during a seismic event could cause loads to exceed allowable limits. The lower containment pressure transmitters are used to automatically actuate the CTS system on a high containment pressure.

The licensee had included the ESRR findings listed above into their restart data base and were in the process of prioritizing and assessing the corrective actions to the these and other findings. While the inspectors did not perform a detailed assessment of the ESRR teams regarding the CTS system, the broad overview did not identify any significant concerns.

The inspectors performed a more detailed oversight of the rotation of the CTS heat exchanger and the problems identified during that rotation. Following the re-installation of the Unit 1, West CTS heat exchanger, the licensee performed routine post maintenance pressure testing. During that testing, the licensee identified several small pressure boundary leaks.

During the assessment of the pressure boundary leaks, several heat exchanger tubes were found to be leaking. The licensee performed eddy-current testing of the Unit 1, West heat exchanger and identified several dozen leaking or thin walled tubes. The eddy-current testing was appropriately expanded to the other three similar CTS heat exchangers. The licensee determined that all four CTS heat exchangers (two per nuclear unit) had flaw indications.

The flaw indications consisted primarily of outside diameter cracks and outside diameter pits. Several tubes were pulled and compared to the eddy-current testing results. Good comparisons were found for the pits; however, the cracks had significant non-conservative discrepancies between the test results and the pulled tubes. The licensee believed that the discrepancies were due to the use of a flat bottom hole standard and began manufacturing an axial notch standard. Prior to the use of the new standard, an industry expert group recommended the use of different frequencies and using new computer models and manipulations got agreement between the eddy-current tests and the pulled tubes within 2 percent.

At the end of this inspection period, the licensee's root cause analysis was continuing. All of the crack indications had been identified in the centerline weld of the tubes. Additional testing and contingency planning were continuing. The licensee had determined that no other safety-related heat exchangers had been procured from the manufacturers that made either the heat exchangers or the tubes used in the heat exchangers.

c. Conclusions

Significant progress had been made by the licensee in the identification of operability issues with the Containment Spray System. At the end of the inspection report period, the system remained inoperable pending resolution of the identified issues. Case Specific Checklist item 10 remained open at the end of the inspection period.

The inspectors observed that engineering personnel provided effective support to maintenance personnel in an effort to resolve the containment spray system problems in a timely manner. As appropriate, the problem scopes were expanded and/or re-assessed to ensure adequate technical resolutions were achieved.

E8 Miscellaneous Engineering Issues

E8.1 Review of Licensee's Readiness for Year 2000

The inspectors conducted an abbreviated review of Year 2000 (Y2K) activities and documentation using Temporary Instruction (TI) 2515/141, "Review of Year 2000 Readiness of Computer Systems at Nuclear Power Plants." The review addressed aspects of Y2K management planning, documentation, implementation planning, initial assessment, detailed assessment, remediation activities, Y2K testing and validation, notification activities, and contingency planning. The inspectors used NEI/NUSMG 97-07, "Nuclear Utility Year 2000 Readiness," and NEI/NUSMG 98-07, "Nuclear Utility Year 2000 Readiness Contingency Planning," as the primary references for this review. The results of the review were discussed with licensee management. The results of this review will be combined with the results of other reviews at other NRC licensees in a summary report to be issued by letter this year.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls (71750)

During normal resident inspection activities, routine observations were conducted in the area of radiological protection and chemistry controls using Inspection Procedure 71750. No uncontrolled releases of radioactive material were identified.

R2 Status of RP&C Facilities and Equipment

R2.1 Plant Equipment Lay-up Program

a. Inspection Scope (71750)

The inspectors reviewed the status of the licensee's lay-up program for plant equipment. The inspectors interviewed Chemistry Department personnel, reviewed condition reports and walked down equipment to assess the adequacy of the plant lay-up program.

b. Observations and Findings

The licensee performed a dual unit shutdown in September 1997, due to operability issues raised during an NRC Architect Engineer inspection. Plant equipment and system chemistry were maintained to support a near term return to power operation. As the return to operation was delayed to address emergency issues, the licensee did not perform a comprehensive lay-up of equipment.

As a consequence of not laying up equipment for the extended outage, evidence of equipment degradation due to corrosion and corrosion products became apparent:

- The Unit 1 west containment spray heat exchanger was hydrostatically tested after pressure boundary welds were made. The heat exchanger hydro failed due to internal leakage later identified as tube leakage. Analysis of pulled tubes by an independent laboratory revealed evidence of microbiologically induced corrosion (MIC) as well as other tube failure mechanisms. A likely cause of the MIC degradation of the tubes was that the equipment was not placed in lay-up during the extended outage (CR 98-7066).

- Failure of heat exchanger tubes in other equipment has occurred during the extended outage of both units. The Unit 1 number 1 reactor coolant pump (RCP) motor cooler (CR 98-8299) and Unit 1 west component cooling water (CCW) heat exchanger (CR 99-0278) failures may be attributed to improper lay-up. Root cause analyses were underway to further refine the cause of these failures.

- In June 1998, failure of tubes in the Unit 2 main turbine lube oil cooler caused a release of 780 gallons of lubricating oil to Lake Michigan. The licensee stated that the probable cause of the tube degradation was the coolers had been left with standing NESW water and the water had aggressively attacked the tube metal. As part of the corrective actions for that failure, the licensee planned to perform an assessment of their equipment lay-up program for long-term shutdown conditions.

In February 1999, an industry organizations assist visit for Chemistry Department reviewed the plant program for lay-up of equipment. The assist team identified that an effective lay-up program for plant equipment did not exist. Chemistry Department personnel developed an action plan to develop procedures and logic trees for placing equipment in lay-up during outages. Equipment was being procured to allow introduction of warm, dry air into secondary systems for corrosion prevention during outages.

The inspectors observed that changes in unit priority and delays in the restart schedule had an adverse impact on placing equipment in lay-up for the extended outage. Licensee senior management made implementation of a comprehensive plant lay-up program a priority.

c. Conclusions

Delay in the placement of plant equipment into an appropriate lay-up condition contributed to the degradation of some plant safety and nonsafety-related equipment. The licensee had recognized the need for an effective lay-up program and had taken action to place equipment in lay-up and to develop planning for lay-up during future outages.

S1 Conduct of Security and Safeguards Activities (71750)

During normal resident inspection activities, routine observations were conducted in the area of security and safeguards activities using Inspection Procedure 71750. No discrepancies were noted.

F1 Control of Fire Protection Activities (71750)

During normal resident inspection activities, routine observations were conducted in the area of fire protection activities using Inspection Procedure 71750. No discrepancies were noted.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on May 27, 1999. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

#R. Antonow, Outage Manager
#J. Arias, Compliance Manager
#C. Bakken, Site Vice President
#P. Barrett, Performance Assurance
#D. Cooper, Plant Manager
#D. Garner, Plant Engineering
#M. Marano, Director Business Services
#T. O'Leary, Radiation Chemistry Manager
#F. Poppell, Regulatory Affairs
#M. Rencheck, Vice President, Nuclear Engineering
#M. Skow, Performance Assurance
#T. Taylor, Regulatory Affairs
#K. VanDyne, Regulatory Affairs
#L. Weber, Operations
#B. Yockey, Performance Assurance
#T. Zemo, Engineering

# Denotes those present at the May 27, 1999, exit meeting.

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Review of LERs
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315/99010-01	NCV	Surveillance procedure did not direct the operators to restore the AES system to a configuration included in the normal operating procedure or enter the appropriate TS limiting condition for operation action statement.
50-315/99010-02	NCV	Nine Unit 1 main steam safety valves had lift setpoints above the Technical Specification allowed values.

Closed

50-315/95002-00	LER	Nine Unit 1 main steam safety valves had lift setpoints above the Technical Specification allowed values.
50-315/97004-00	LER	Main steam safety valve exceeds allowable lift setpoint due to setpoint drift.
50-315/98021-03	EEI	Surveillance procedure did not direct the operators to restore the AES system to a configuration included in the normal operating procedure or enter the appropriate TS limiting condition for operation action statement.
50-315/99010-01	NCV	Surveillance procedure did not direct the operators to restore the AES system to a configuration included in the normal operating procedure or enter the appropriate TS limiting condition for operation action statement.
50-315/99010-02	NCV	Nine Unit 1 main steam safety valves had lift setpoints above the TS allowed values.

LIST OF ACRONYMS

ASM	Assistant Shift Manager
CCP	Centrifugal Charging Pump
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CSC	Case Specific Checklist
CTS	Containment Spray System
CVCS	Chemical and Volume Control System
DRP	Division of Reactor Projects
ESRR	Expanded System Readiness Review
ESW	Essential Service Water
I & C	Instrumentation and Controls
IHP	Instrument Head Procedure
IMP	Instrument Maintenance Procedure
JO	Job Order
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MC	Manual Chapter
MHP	Maintenance Head Procedure
MOV	Motor Operated Valve
MSSV	Main Steam Safety Valve
N/A	Not Applicable
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OHI	Operations Head Instruction
OHP	Operations Head Procedure
PMI	Plant Manager's Instruction
PMP	Plant Manager's Procedure
PMSO	Plant Manager's Standing Order
PMT	Post Maintenance Testing
PPA	Plant Performance Assurance
PDR	Public Document Room
PORV	Power Operated Relief Valve
STP	Surveillance Test Procedure
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation